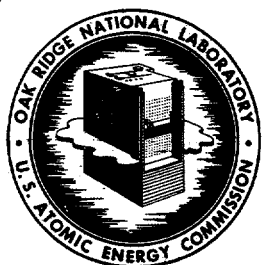


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PRELIMINARY REPORT
OPERATION OF THE AIRCRAFT REACTOR EXPERIMENT

Work Performed by
Aircraft Nuclear Propulsion Project

Report Written by
J. L. Meem and Wm. B. Cottrell

Foreword: The following report was written specifically for publication as a technical note in the next (February) issue of the Reactor Science and Technology Journal. Distribution of the report as a CF memorandum is for the edification of those close to the project who have need of the information in advance of the final detailed report.

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3

PRELIMINARY REPORT
OPERATION OF THE AIRCRAFT REACTOR EXPERIMENT

A. Introduction

The Aircraft Reactor Experiment, the construction of which was the subject of a previous article in this journal,¹ recently concluded its scheduled operation. This note is a brief description of the operation and results obtained from the experiment.

The primary object of the experiment was to build and operate a high-temperature, low-power, circulating-fuel reactor using materials which would be amenable to a high-power aircraft-type reactor. The fuel selected for the experimental reactor was a mixture of the fluorides of sodium, zirconium, and uranium; Inconel was employed both as the fuel container and structural metal. The operating fuel temperatures were to range as high as 1500°F, with up to 350°F temperature rise across the reactor. The design power was one megawatt and the operation was scheduled to be terminated after one hundred megawatt hours had been accumulated.

A secondary objective was to obtain as much experimental data as possible on the reactor operation - the values of critical mass and the fuel temperature coefficient were of particular interest. The value of the critical mass was obtained automatically in taking the reactor critical. The measurement of the fuel temperature coefficient was obtained but only after calibrating the regulating rod. With a calibrated regulating rod, a number of other interesting experiments were readily conducted.

At the time the concept of an aircraft reactor experiment was accepted in the summer of 1950, it was to be a prototype of a potential aircraft power plant. Since that time, however, the evolutionary processes which eventually resulted in the actual ARE were of such consequence that the experimental reactor did not, nor was it intended to, resemble an aircraft power plant. Rather it was to illustrate the principle that a high-temperature, circulating-fuel reactor could be constructed, taken critical, and operated in the megawatt range.

B. Description

The Aircraft Reactor Experiment consisted essentially of the circulating-fuel reactor and the associated pumps, heat transfer equipment,

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1. Building Reactors - Aircraft Reactor Experiment, Reactor Science and Technology, TID-2013, Vol. 4 - No. 2, June 1954.

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controls, and instrumentation required for its safe operation. A schematic arrangement of the reactor system is shown in Fig. 1. The major functional parts of the system are discussed separately below.

The reactor assembly consisted of an Inconel pressure shell in which beryllium oxide moderator and reflector blocks were stacked and through which passed fuel tubes, reflector cooling tubes, and control assemblies. Elevation and plan sections of the reactor are shown in Fig. 2 and Fig. 3. The innermost region of the lattice was the core, which was a cylinder approximately 3 ft in diameter and 3 ft long. The core was divided into six 60-deg sectors, each of which included one serpentine fuel-tube coil that passes through 11 stacks in series, as illustrated. The six serpentine coils were connected in parallel by means of external manifolds.

A reflector with a nominal thickness of 7.5 in. was located between the pressure shell and the cylindrical surface of the core. The reflector consisted of beryllium oxide blocks similar to the moderator blocks but with 0.5-in. holes for the passage of the reflector coolant. The reflector coolant - sodium - was admitted at one end of the pressure shell, passed through the reflector, bathed the pressure shell, filled the moderator interstices, and exited at the other end of the pressure shell.

Circulation of both the fuel and sodium was effected by vertical shaft, gas-seal centrifugal pumps, with which flow rates up to 84 gpm could be obtained. The reactor heat was abstracted from the circulating fuel outside the core by means of two fuel-to-helium heat exchangers through which the fuel was circulated. The helium was then cooled by passing it through two helium-to-water heat exchangers, and the hot water was discharged. The sodium was cooled by a comparable sodium-to-helium-to-water heat exchange system.

Helium flow rates in the fuel-to-helium heat exchanger were controlled by a variable-speed electric motor that drove the helium blower. Control of the helium flow rate in this manner permitted smooth control of reactor power at any reactor temperature for which the nuclear controls were set, within the capacity of the heat removal system.

The relatively high melting point of the circulating fuel (around 1000°F for the NaF-ZrF₄-UF₄ fuel with a composition in mole per cent of 53.2-40.5-6.3) required that all equipment within which this coolant was circulated be heated to permit loading, unloading, and low-power operation. This heating was accomplished by means of electrical heaters attached to all components of the fuel and sodium systems, that is, pressure shell, heat exchanger, pumps, tanks, as well as all fuel and sodium piping.

The fuel carrier, NaF-ZrF₄ (50-50 mole %), was first loaded into tanks provided in a shielded pit adjacent to the reactor and heat exchanger pits. This molten salt was then forced into the fuel system by helium

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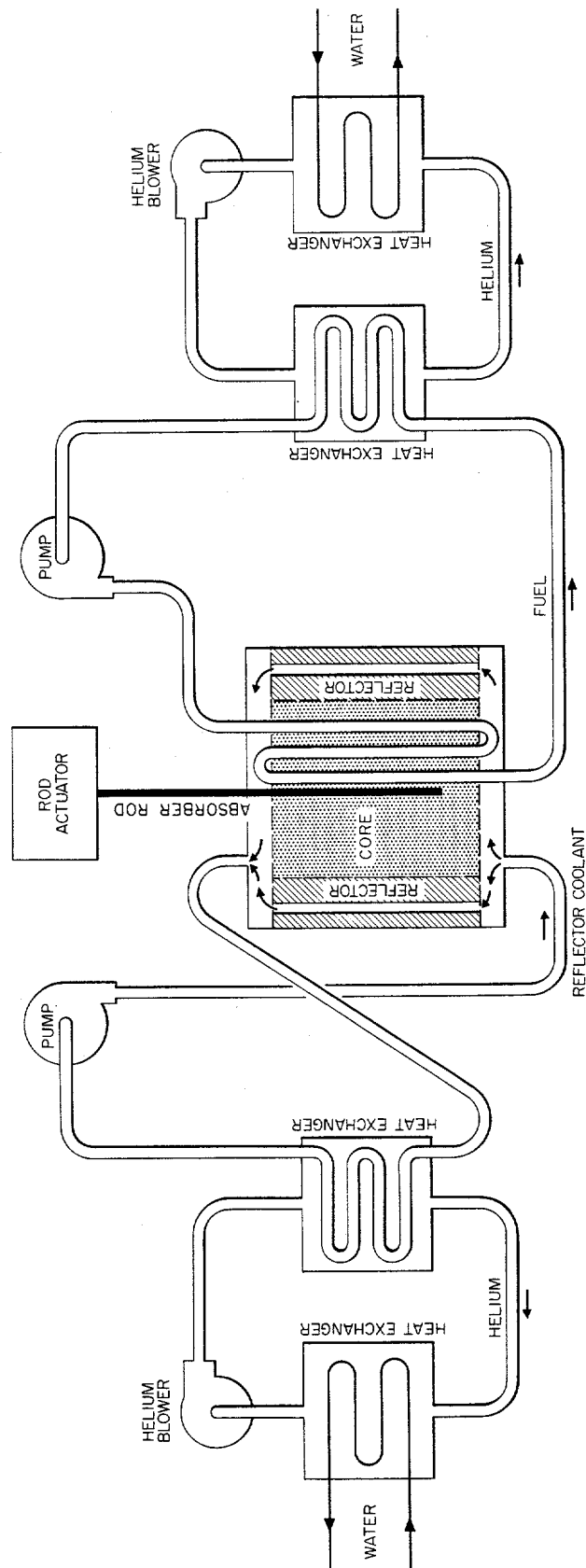


Fig. 1. Schematic of Aircraft Reactor Experiment.

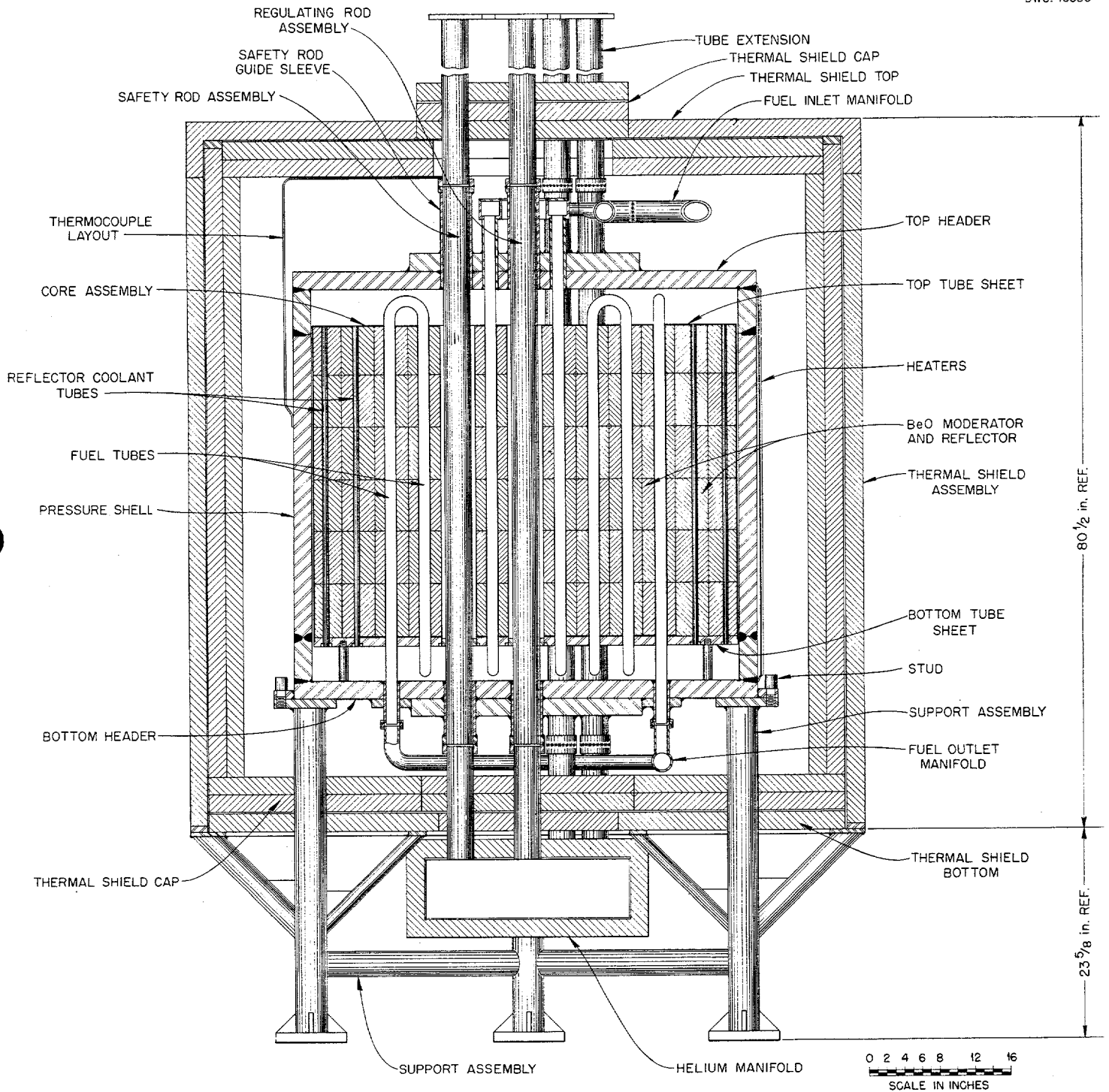


Fig. 2. The Reactor (Elevation Section)

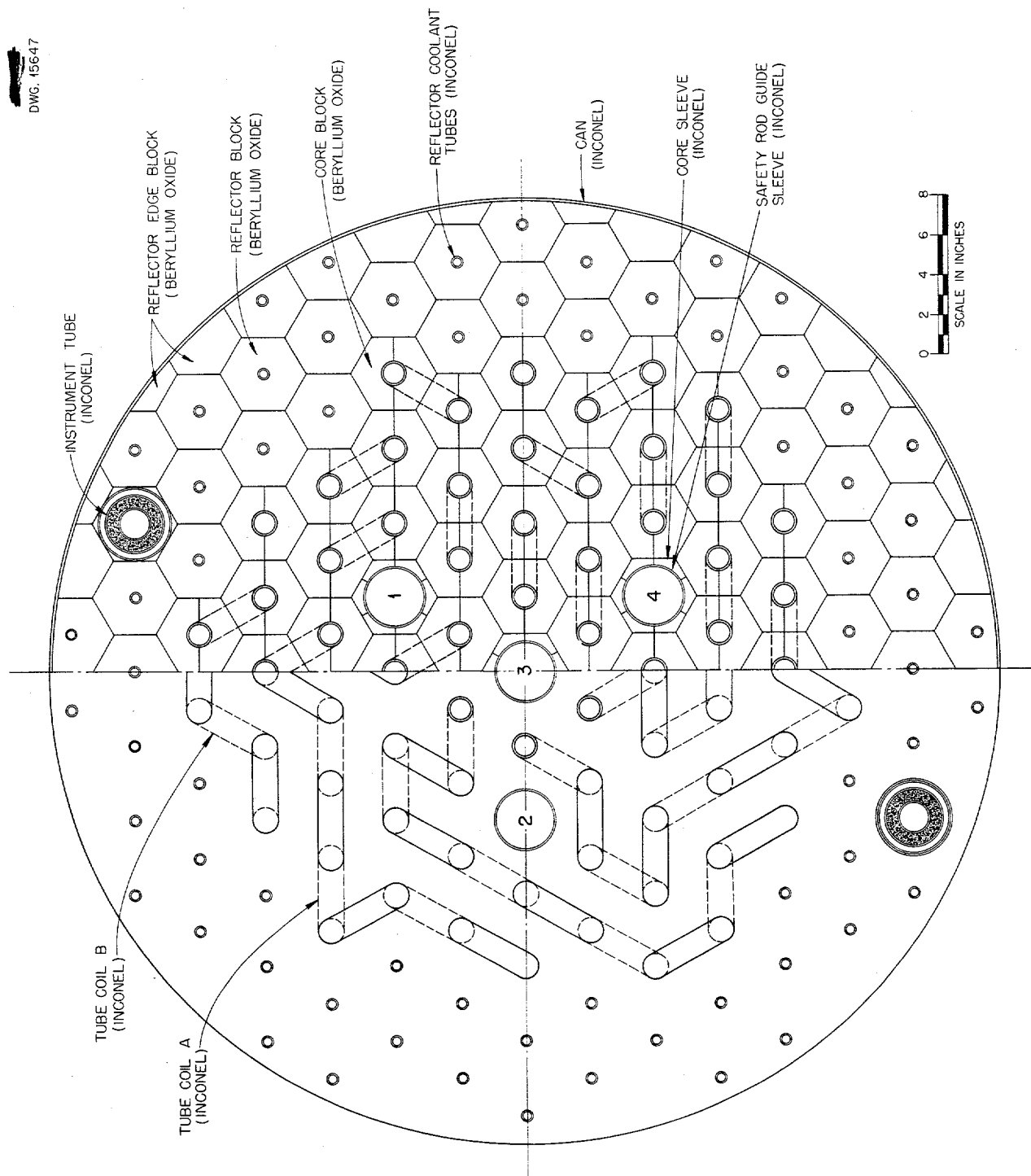


Fig. 3. The Reactor (Plan Section).

pressure. The uranium-bearing fuel component, NaF-UF_4 (66.7-33.3 mole %), was then added by the fuel-enrichment system.

C. Critical Experiment

In order to make the reactor critical, uranium in the form of molten Na_2UF_6 was added to the barren carrier - NaZrF_5 - with which the fuel system was initially filled. The phase diagram of the ternary fluoride system $\text{NaF-ZrF}_4\text{-UF}_4$ is such that there are no intermediate mixtures between Na_2UF_6 and NaZrF_5 with melting points higher than 1185°F - the melting point of the concentrate. Addition of the concentrate increased the melting point of the fluoride system above that of the carrier, i.e., 955°F . The fuel composition at initial criticality was 52.8-41.5-5.7 mole % ($\text{NaF-ZrF}_4\text{-UF}_4$) which has a melting point of 990°F , while the final fuel composition (which included excess uranium) had a composition of 53.2-40.5-6.3 mole % ($\text{NaF-ZrF}_4\text{-UF}_4$) and a melting point of 1000°F .

It was initially intended to remotely add the concentrate to the fuel system from a large tank which contained all the concentrate, after first passing it through an intermediate transfer tank. This system was discarded when temperature control and continuous weight measuring instrumentation on the transfer tank proved to be unsatisfactory. Instead a less elaborate, but more direct method of concentrate addition, was employed. This enrichment operation involved the successive connection of numerous small concentrate containers to an intermediate transfer pot, which was in turn connected to the fuel system by a line which injected the concentrate in the pump above the liquid level. Each of the concentrate containers was weighed before and after in order to determine the amount of uranium injected into the system. The concentrate was first batched down into various cans containing from 2 lb of Na_2UF_6 (for rod calibration) up to 33 lb (as was used during the first subcritical loadings). In this enrichment operation the pump bowl served as a mixing chamber and uniformly distributed the concentrate into the circulating stream.

While the first concentrate addition was made on October 30, the reactor did not become critical until three days later (3:45 P.M., Nov. 3). However, most of the intervening time was expended in clearing the end of the transfer line at the pump which, because of limitations inherent to only this particular design, was difficult to heat and even more difficult to service.

The approach to criticality was carefully charted, after each fuel addition. The resultant curve of reactivity ($1 - \text{multiplication}$) as a function of the addition of fuel (in terms of lb of U-235 per cu.ft.) is given in Fig. 4. The data from three different ionization chambers is presented; "Meters #1 and #2" were fission chambers located in the reflector, and the BF_3 counter was located external to the reactor at midplane of the

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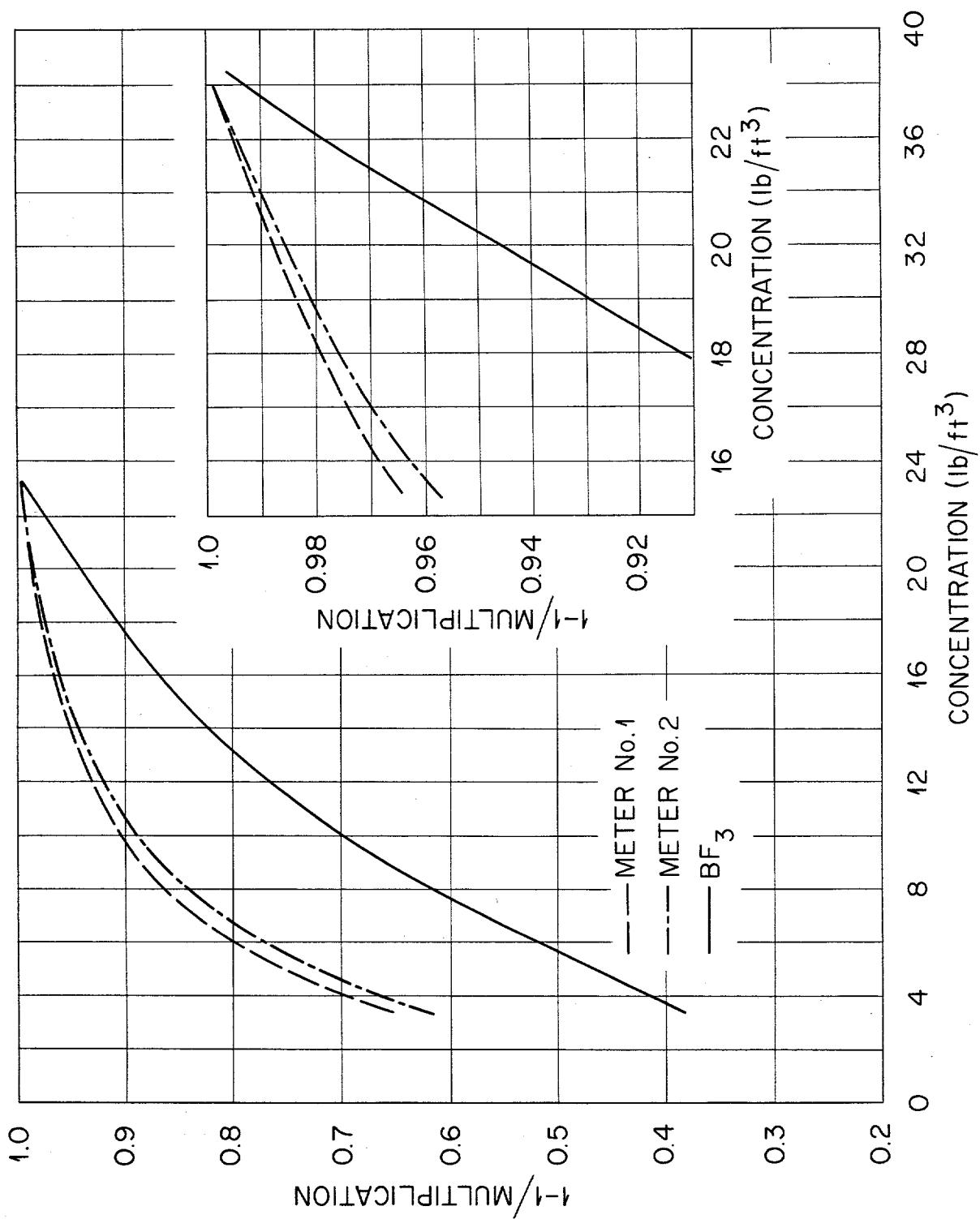


Fig. 4. Approach to Criticality.

cylindrical side. The unique shape of these curves (which when first extrapolated suggested a much lower critical mass than was actually required) is believed to be due to the particular radial flux distribution of the reactor at the location of the chambers and the change in this distribution as criticality is approached.

The calculated volume of the carrier in the fuel system (before concentrate addition) was 4.82 cu.ft. (The only significant check on this value was obtained from subsequent analyses of fuel samples together with the known amounts of concentrate added.) While the total amount of uranium (U-235) added to the system in order to make the reactor critical was ~135 lb, because of the amounts withdrawn from the system for samples and in trimming the pump level the uranium concentration at criticality was 23.7 lb/cu.ft.; or since the calculated volume of the 1300°F core was 1.37 cu.ft., the "cold" clean critical mass of the reactor was 32.5 lb of U-235.

D. Low Power Experiments

There are several "experiments" which were performed on the critical reactor at low power, including reactor power and rod calibrations. In addition, the effects of the process system parameters on reactivity were noted and a preliminary measurement of the temperature coefficient undertaken. The tests commenced during the morning of November 4 and were completed by noon on November 8.

The reactor had four control rods (see Fig. 2). The regulating rod (for fine control) was located on the axis of the cylindrical core and three safety rods (for coarse control and to scram the reactor) were symmetrically located on a 7.5" radius about the regulating rod. The regulating rod "poison" was obtained from a stainless steel tube, whereas the safety rods were fabricated from annular sections of hot-pressed B₄C.

The regulating rod was calibrated both by the addition of fuel as well as the resultant pile period upon withdrawing the rod (as derived from the Inhour equation). The value of the rod was first obtained by noting the amount of rod insertion required to maintain a constant power level as a finite amount of fuel was added to the system. This information together with a calculated value of the mass reactivity coefficient ($\Delta k/k \cdot \Delta m/m$) equal to 0.232 (subsequently validated by period calibration) permitted a determination of the value of the rod. The technique of rod calibration by reactor period was employed both at design fuel flow (48 gpm) and with no fuel flow. The data from each of these tests is presented in Fig. 5. Although there is considerable scatter, the data from the different rod calibration techniques appear to be mutually confirmatory and the rod value of 0.032 $\Delta k/in.$ as obtained from this calibration was used throughout the remainder of the experiment. It should be noted, however, that the period calibration with no flow is

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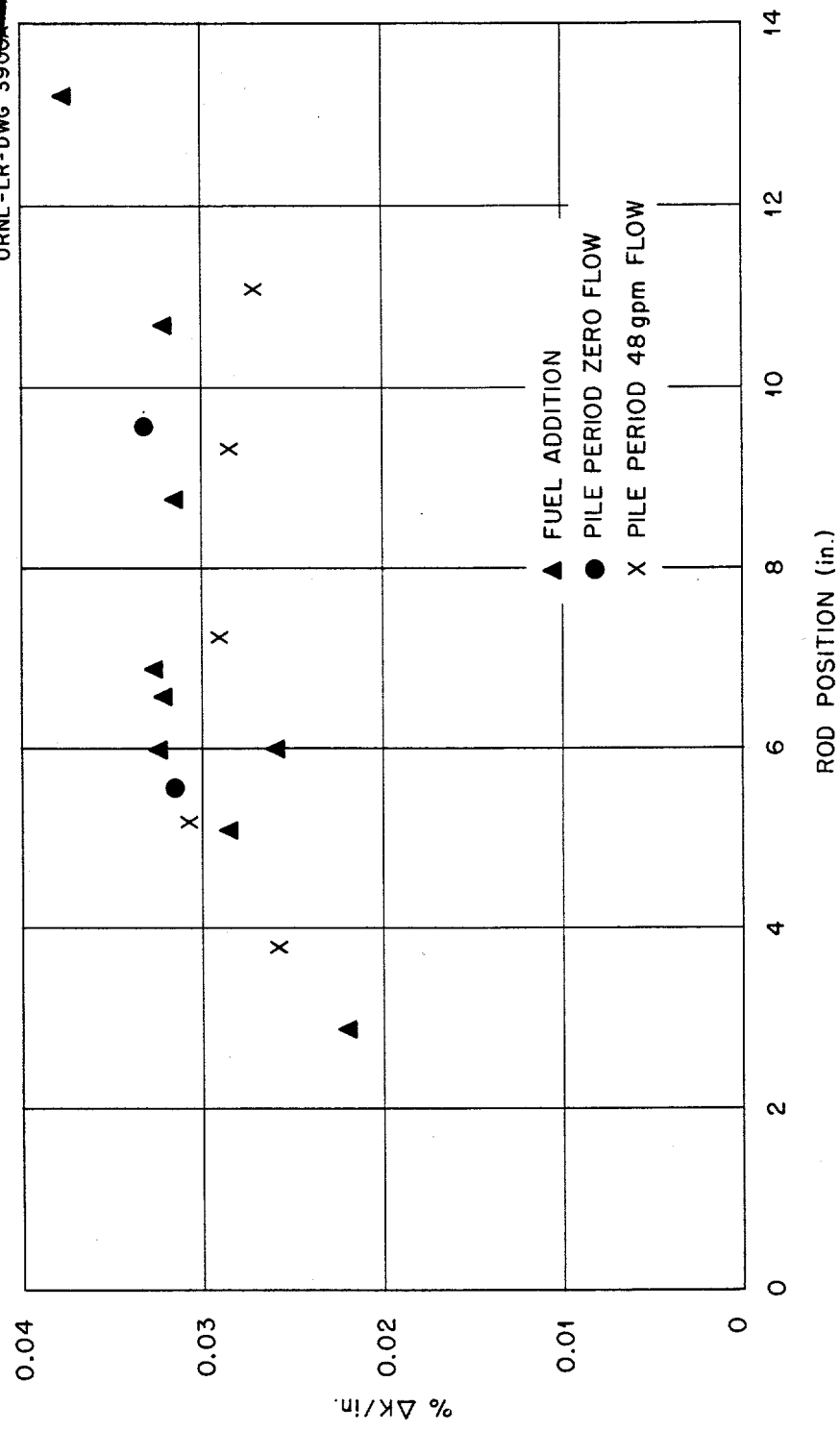


Fig. 5. Calibration of Regulating Rod.

believed to give the best data since the Inhour equation is applicable without correction and the resultant value of the rod is not dependent upon a reactivity coefficient.

While the reactor power was first estimated from the fission chamber counting rate, attempts were made to confirm this estimate by operating the clean reactor at some low power for a one-hour period, then withdrawing a fuel sample and taking a count of the sample. This calibration was attempted first at an estimated one watt and then at ten watts. The fuel activity from this one watt hour run was too low to count accurately, but that from the ten watt hour run indicated a power was 13.5 watts. The nuclear instrumentation was then calibrated on this basis, although it subsequently developed that most all the volatile as well as gaseous fission products were apparently continuously removed from the fuel at the pump. Consequently, the actual power was probably much greater than that indicated by the fuel sample.

Attempts were made to measure the temperature coefficient both when the reactor was subcritical and again during the low power operation. In both instances, it was established that the coefficient was negative and in the latter case it was determined that the magnitude was approximately $5 \times 10^{-5} \Delta k / ^\circ F$. A more accurate determination of the magnitude of the temperature coefficient was deferred until the high power runs.

As a part of the low power operation, the shim rods were calibrated in terms of the regulating rod. Each of the three shim rods had $\sim .15\% \Delta k$ per inch for most of their 36 inches of travel.

E. High Power Experiments

The reactor was finally taken to full power (an estimated 1 Mw) at 6:20 P.M., November 9, some six days after it first became critical. This power level was attained after a 30-hr period of intermittent operation during which there were periods of operation with power levels of 10 kw, 100 kw, 500 kw, and finally 1 Mw. Power was attained as anticipated merely by increasing the speed of the blower which cooled the fuel heat exchanger. Full power could have been obtained at once except for the natural tendency to proceed slowly into such an unexplored regime - a fortunate decision since there was some leakage of gaseous activity from the vent system into the pit and subsequently into building atmosphere. Further difficulty from this source was circumvented by operating the pit at subatmospheric pressures and remotely exhausting the pit gases to the atmosphere.

Once power was attained the reactor was operated intermittently, and at various power levels during the next several days as required to complete the desired tests. These tests included measurement of the temperature coefficient of reactivity, a power calibration from the process instrumentation, a determination of the effect of large increases in reactivity, and

were concluded by a 25-hr run at full power to see if there was any detectable buildup of xenon.

The temperature coefficient of reactivity was determined simply by placing the regulating rod on the flux servo and then increasing the speed of the blower cooling the fuel. The change of rod position (converted into reactivity) divided by the change in the reactor mean temperature determines the reactor temperature coefficient. The absolute value of this coefficient was initially quite large, decreasing after two minutes to a relatively constant value of $-5.5 \cdot 10^{-5} \% \Delta k/^\circ F$. Further analyses of the data is under way to ascertain the precise value of the instantaneous fuel temperature coefficient which is, of course, the most important characteristic effecting the control of a power reactor. It is certain that this coefficient was considerably larger than expected, and that the reactor itself was exceptionally stable.

In this, as in any potential power reactor, the reactor behavior as a result of large increases in either reactivity and power demand are of particular interest. With a circulating fuel reactor operating at power, insertion of the safety rods reduces the reactor mean temperature. The power level, on the other hand, is controlled by the rate at which heat is withdrawn from the circulating fuel. The effects of several of these operations are illustrated graphically in Fig. 6 which records the reactor inlet and outlet fuel temperatures over a 100-minute test. Shown on this figure are the readings of the twelve thermocouple readings on the inlet and outlet of each of the six parallel fuel circuits through the reactor. Starting at the right with a mean temperature of $\sim 1320^\circ F$, the fuel helium blower speed was gradually increased over a 12-minute period until a reactor ΔT of $250^\circ F$ was obtained. (This ΔT , as will be shown later, corresponds to a reactor power of ~ 2.5 Mw.) After some 18 minutes at power the helium blower motor was turned off and the ΔT of $\sim 0^\circ F$ was restored in about 8 minutes. However, by this time the reactor mean temperature was $1380^\circ F$ and was therefore decreased (to $1350^\circ F$) by inserting the shim rods. The reactor was again brought to power in two minutes (~~starting at 2353~~) and after 8 minutes at power the shim rods were withdrawn to increase the mean temperature. It is most significant that during the 2-minute interval required to bring the reactor from a nominal power of 100 kw to 2.5 Mw the shortest recorded period was only 14 seconds! Furthermore, when at power group withdrawal of the shim rods ($0.02\% \Delta k/sec$) resulted in a period of only 10 seconds! These two limiting periods were consistently reproducible.

Additional insight on the behavior of the negative temperature coefficient is also afforded by Fig. 6. By time 84 min. the blower was off and the reactor power was reduced to ~ 100 kw. The shim rods were then inserted to make the reactor subcritical. At 86 min the fuel helium blower was turned on. The higher-density cooled-fuel made the reactor critical and in 2 minutes a

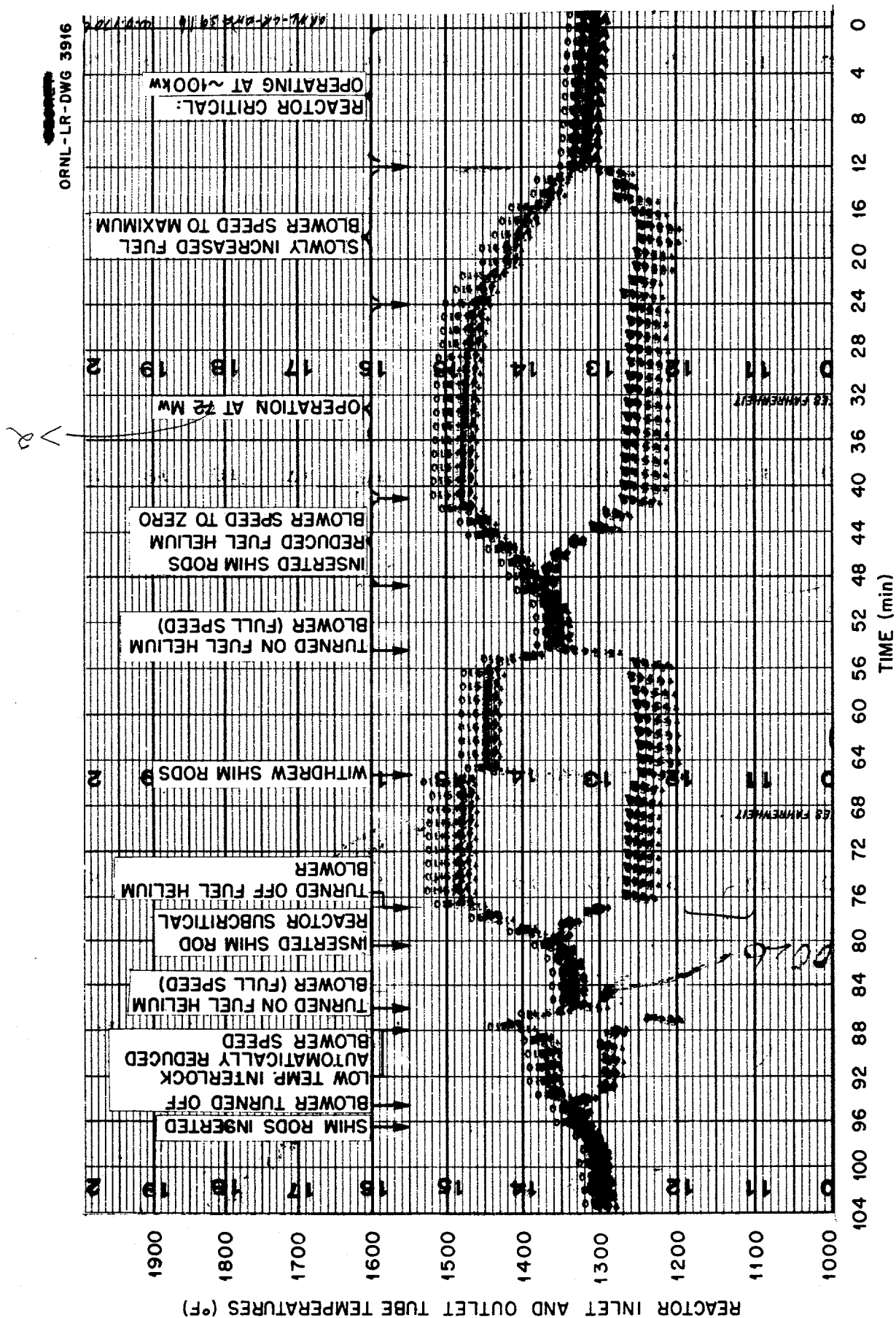


Fig. 6. Power Excursions.

ΔT of $\sim 200^\circ\text{F}$ was obtained. At this time, however, the blower speed was automatically reduced by a low temperature signal. The blowers were subsequently shut off and the shims inserted.

Although the reactor power level had been calibrated against the activity count of a fuel sample, the actual reactor power remained in doubt throughout the experiment, not only because of uncertainty regarding the retention of fission products by the fuel but also because of discrepancies in heat balances in the process systems, i.e., heat removed from fuel and sodium vs heat picked up in their water heat dumps. While the causes of these discrepancies are now being analyzed in detail, the most reliable estimate of the reactor power is believed to be that obtained from the ΔT 's and flows in the fuel and sodium system themselves. During one typical period of power operation the fuel ΔT of 370°F at 45 gpm accounted for > 1.9 Mw in the fuel, while the sodium ΔT of 115°F at 150 gpm accounted for another > 0.6 Mw in the sodium, resulting in a total reactor power in excess of 2.5 Mw. The ΔT 's quoted above were obtained from the fuel pipe temperatures to and from the reactor at the time the reactor inlet and outlet tube temperatures (as in Fig. 6) recorded only at 250°F ΔT . While there are several possible phenomena which could contribute to these low tube temperatures (external cooling, surface layers, radiation, etc.), there are some 10 to 15 couples on the inlet and outlet pipes which gave consistent readings. The outlet pipe during the 2.5 Mw operation mentioned above ran at equilibrium temperature of 1580°F , and was during transients in excess of 1600°F .

The last scheduled experiment to be conducted on the reactor was a measurement of the xenon buildup during a 25-hr run at full power. The amount of xenon buildup was observed by the amount of regulating rod which had to be withdrawn in order to maintain a constant power level. However, during the 25-hr power run the regulating rod had only been withdrawn 0.3 in. or one-thirtieth of the amount calculated had the xenon remained in the fuel. While the question of the xenon concentration in the fuel will not be finally resolved until it is possible to get and analyze a fuel sample, it now appears that most all fission fragments - volatile as well as gaseous - were removed from the fuel by the purging action of the pump, since the activity of the dumped fuel is considerably less than that calculated for the fuel assuming it retained all its fission activity.

The scheduled tests above were completed by 8:00 A.M., November 12. The reactor operation was then demonstrated for all those who attended the ANP Information Meeting on Friday, November 12. During the 12-hr period starting 8:00 A.M. Friday until 8:00 that evening, the reactor was brought to full power and back down some 21 times. The resulting temperature cycling was probably as severe as that to which an aircraft reactor would be subject. At 8:00 P.M. November 12, having completed the scheduled experimental program and having logged over 100 Mw hours, the reactor was

shut down for the last time. The following morning (after allowing both the fuel and sodium to circulate over night) the fuel and then the sodium were dumped into their respective dump tanks.

F. Conclusions

The Aircraft Reactor Experiment was operated successfully for over one hundred megawatt hours. During this time the maximum equilibrium outlet fuel temperature was 1580°F and the maximum power level was over 2.5 Mw. The temperature coefficient of reactivity two minutes after the introduction of a perturbation was -5.5×10^{-5} , although its instantaneous value was considerably higher. As a result, the reactor was extremely stable while operating at power and there were no control problems. The critical mass was 32.5 lb of U-235, in reasonable agreement with the calculated value of 30 lb. Operation of the process systems and their instrumentation were exceptionally trouble-free during the six days of operation from the time the equipment was sealed in the pits until the sodium and fuel were dumped.

It is believed that most of the fission product gases evolve from the liquid fuel. In a 25-hr run it was determined that no more than one part in 30 of the xenon poison remained in the fuel. Furthermore, preliminary radiation measurements of the dumped fuel show it considerably less radioactive than anticipated. Although this will vastly simplify fuel recovery and equipment salvage problems, the implications to a high-power aircraft design and operation are of even greater consequence.